Technical University of Denmark

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**Abstract**

The work of the Department of Reactor Technology within the following fields is described:

Reactor Engineering
Steel pressure vessel research
Reliability
Reactor physics
Steady-state thermohydraulics
Accident analysis
Containment analysis
Experimental heat transfer
Core dynamics and power plant simulators
Experimental activation measurements and neutron radiography at the DR 1 reactor.


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## 1. Introduction

The Department of Reactor Technology comprises six sections:

Reactor Engineering
Reactor Physics
Heat Transfer and Hydraulics
Experimental Heat Transfer
Reactor Dynamics
The DR 1 reactor

The work of these six sections during 1975 is described in this paper.

Activities predominantly concern research and development on a short- to medium-term time-scale (1-4 years), but some tasks are carried out on a short-term basis for outside bodies. In addition the general collection of knowledge of reactor systems, norms and safety criteria can be considered short-term work.

During the first half of the year, the Department took part in consultations with the Danish Inspectorate of Nuclear Installations concerning the work, or the exercises and preparations for this work, involved in the licensing procedure for nuclear power stations. One activity resulting from these consultations was a containment exercise, where the licensing procedure for the containment system of a GE-BWR was tested by going through GESSAR together with all the appropriate USNRC Standard Review Plans, Regulatory Guides, etc. This exercise was to be carried out within two and a half months' time, and involved seven of the Department's graduate staff and three from other Risø departments. These staff went through several thousand pages of GESSAR and SRP's, etc., ending up with a greater knowledge of the production of paper by the USNRC, but also more sceptical of the applicability of American licensing procedure in a country without the experience, background information, and manpower as the USNRC.

Towards the end of the year another safety exercise was undertaken in collaboration with the Risø Electronics Department. This study contains two parts.; the first is a study of the logic structure of the safety criteria, describing permissible doses and design basis accidents. The second part initially tracks the systems and circuits involved in an accident where all off-site power is lost,

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whereafter the consequences of this accident are to be studied.

Simultaneously with these two exercises, the Section of Reactor Engineering initiated systematic work on system descriptions and documentation of important systems. Each system is described in a report containing information on design, construction, function and norms. This work is based on the Safety Analysis Reports supplied by vendors, and is considered to constitute background material for the assistance to be given to the Inspectorate at a later date.

Thus preparations for assisting the Inspectorate have in several ways introduced short-term work affecting (medium-term) research on the analysis of reactor operation and safety.

In the field of medium-term safety research, the major event of the year was the negotiations on the IEA-sponsored reactor safety research projects. So far negotiations have mostly affected managers of safety research, but it is hoped that the actual work will start at the beginning of the new year. It is the intention of the Department to participate in the American LOFT, PBF and BSST experiments, and one of the main activities of the Section of Heat Transfer and Hydraulics will be through the Nordic NORHAV project, to create codes and produce calculation results for the loss-of-coolant accident analyses of LOFT and PBF.

Work was completed on blow-down and containment experiments in the Marviken I series, in Sweden, at the beginning of the year, and the Department now participates in the new Marviken II containment response test series, where pressure oscillations are studied.

Of task more related to outside affairs, a study was made of the economics of atomic power plants in the Danish grid, with the possibility of supply from neighbouring systems, while one member of staff participated in a ministerial working group on power plant economics.

Finally, members of the Department have lectured at the universities on reactor technology and physics, and seven post-graduate students are working on lic.tech. (Ph.D.) theses within the subject fields of the Department.

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## 2. Section of Reactor Engineering

### 2.0 Introduction

The main object of work in the section of reactor engineering is to establish and maintain competence in overall nuclear power plant design and evaluation.

In addition, the section conducts or participates in activities of a more specific nature in partial support of the main theme. Collaboration takes place with other sections in the Department of Reactor Technology, with other departments at Risø, and with outside bodies, including commercial enterprises. Specific topics being treated at present are reliability analysis, analysis of pressure vessel safety, fuel element development, prediction of radiation doses to personnel in nuclear power plants, and analysis of the failure probability of nuclear fuel elements.

The competence and know-how established in the section is intended for application in safety evaluations and similar investigations aimed at analyzing nuclear power plants in areas of interest to governmental authorities, utilities, or the community as a whole. It is also aimed to disseminate knowledge of the various aspects of nuclear power both to specialist groups of professionals and students and to the general public.

### 2.1 Steel (Pressure Vessel) Research

The purpose of the work is to gain knowledge sufficient to evaluate the fabrication, performance and reliability of steel components used in light water reactors. Within this scope, work concentrates on the following two areas:

a. The development of methods for estimating the reliability of steel components, especially the pressure vessel (probabilistic fracture mechanics)

b. Follow up of developments in steel pressure vessel technology and code requirements for such components.

The main effort has been concentrated on the assessment of the reliability of the pressure vessel based on probabilistic fracture mechanics. Failure probability is calculated by two computer codes utilising the Monte Carlo method. The first code, PFM 690, calculates the crack growth as a function of time on the basis of an initial crack distribution, crack growth characteristics and stress transients. This code utilises simple Monte Carlo. The second code PEP 706 (a new version of PFM 683) calculates the probability of failure based on a stress-strength model and uses Monte Carlo with importance sampling.

The new crack growth data from the HSST-program and others conducted at very low frequencies (1 cpm or lower) shows such an increased crack growth that the Paris formula, which has been used hitherto, now becomes mathematically unstable. This fact seems not only to demand a much better statistical data basis, but also to require another model.

The possibility of using Beta-functions for the parameters was intensively investigated because of their special features (i.e. a four-parameter probability density function (p.d.f.) with an upper and lower bound and two shape parameters). Giving the p.d.f.'s for all the parameters and their weighting functions the same mathematical form should make it possible to automate the selection of the weighting functions for importance sampling.

A new computer program, PHE2, was developed for calculation of the probability of fracture of components. The program is of the analytic type, in contrast to the Monte Carlo program PEP 706. The calculations performed have shown good agreement with previous calculations using PEP 706.

The new program works much faster than the previous one; thus a computation lasting ten minutes with PEP 706 can be performed in only 15 seconds using PHE2. This feature is, of course, particularly important for analyses involving series of calculations such as parametric studies. However, the analytical approach can only be applied to straight-forward problems. The probability density functions of the parameters in the stress intensity factors can be determined either pointwise or by means of Weibull- or Beta-functions. In the present version of PHE2, however, the number of different parameters in the stress intensity factors is limited to two; but this has been of no practical importance in the calculations performed so far.

The number of international contacts and collaboration projects was extended. First, the two computer codes PFM 690 and PEP 706 were



handed over to a British firm; running in has been successfully completed. Collaboration was initiated with Babcock and Wilcox, Lynchburg, U.S.A., concerning the probabilistic aspects of neutron embrittlement for reactor vessels. Moreover contact with Westinghouse Research Laboratories, Pittsburg, U.S.A., was further developed. This collaboration will concern the development of a better statistical basis for the p.d.f. of the critical stress intensity factor and an evaluation of the uncertainties in determining the stresses, i.e. the stress intensity factor.

## 2.2 Reliability

The purpose of this work is to keep up to date the sections knowledge of and ability within the field of reliability, seen in the light of the increasing demand for reliability analysis both for the design of systems and for the safety analysis of nuclear power stations.

In accordance with the schedule for 1975, work has comprised the following three subjects:

I. General reliability techniques

II. Probabilistic fracture mechanics (descriped in 2.1 above)

III. Comparison between WASH 1400 and SUSS (Swedish Urban Siting Study).

### I. General reliability techniques

The computer program REDIS for system reliability analysis (see Annual Progress Report for 1974) was further developed. It is now possible to choose freely between the following types of probability density functions for the times to failure and the repair times of each individual fault: exponential, Weibull, normal, Log normal and constant. A new test model was incorporated for routine testing of arbitrarily selected groups of components at specified intervals of time. Further, a histogram procedure was incorporated for plotting of parameters wherever desired.

The REDIS program was described in a paper, IAEA-SM-195/17, presented at the International Symposium on the Reliability of Nuclear Power Plants held at Innsbruck, April 14-18. In addition, a detailed description and user's manual for the code was prepared

(Risø-M-1781).

A masters thesis dealt with reliability techniques, comprising a comparison between calculations with the REDIS program and analytical calculations. After correction of an error in the code, which would be important under extreme conditions, there was complete agreement between the two types of calculation in all cases.

III. Comparison between WASH 1400 and SUSS

In coorperation with the Electronics Department, the section contributed to a Nordic comparison between WASH 1400 and the corresponding Swedish report SUSS (Närforläggningsrapporten).

Much greater efforts were put into WASH 1400 than into SUSS and of course, this influences them. On many points the former report is more realistic, since it is based on original work, while the latter is based on information from the literature. However, the final conclusions are consistent indicating that nuclear power plants do not involve greater risks for society than several already existing risks. The comparison is descriped in AB Atomenergi S-483.

## 2.3 Fuel Element Development

This work is part of a programme carried out by the Metallurgy Department. The Department of Reactor Technology generally participates in the design of fuel elements and, in connection with post-irradiation examinations in studies of the mechanical interaction between $UO_2$-pellets and cladding. Two fuel rods (HP 060 and 061) irradiated in DR 3 were examined.

## 2.4 Radiation Doses in Nuclear Power Plants

This project was started up in February 1974 with the aim of providing a mathematical model for calculation of the radiation fields in various locations in a nuclear power plant and of the radiation doses received by the personnel of the plant. A power plant with a boiling water reactor was chosen for the study.

During 1975 work concentrated on refining and extending the computer programs previously made, and on setting up a code for shielding calculations.

The programs for calculation of the time-dependent radioactive



inventory of the coolant circuit were provided with options that include shut-down periods in the simulation of the power history of the plant. During these shut-down periods a possibility exists for simulating decontaminations of parts of the circuit and for the exchange of components, e.g. filters and fuel. Furthermore, subroutines were included that make it possible to plot the results on a Calcom-plotter. The program calculating the fission product inventory was altered with respect to the solution of the rather large system of differential equations. Previously a quite sophisticated predictor-corrector method was used for this purpose; but it proved more time-consuming than desirable. Therefore a more simple method - a so-called "Modified Euler" - was set up. It has given a considerable reduction in computer time without introducing significant numerical errors. A comparison between results from calculations with the programs and data from the literature suggests that the programs work satisfactorily.

A code for shielding calculations was completed. It is quite simple, using a point kernel technique with build-up. The source geometry is cylindrical, but a possibility exists for changing this into a conical form (turbines). The shielding may be a slab or a cylinder concentric with the source; several different layers of shielding can be accounted for. The build-up factors for some common materials are stored as polynomials in photon-energy and material thickness.

So far, tools were made for a calculation of radiation field. Future work will combine the calculations of radioactive inventories with shielding calculations to obtain the radiation fields outside the components of the coolant circuit. One major problem that could arise in this connection is the acquisition of correct data for materials etc., in order to make a realistic calculation.

## 2.5 Reliability of Fuel Cladding

This project is part of work in connection with a doctoral dissertation and conducted in collaboration with the Metallurgy Department.

Up till now work was concerned with the development of a simplified model for cladding/pellet behaviour under reactor conditions. A certain amount of material data for Zircaloy was collected.

The simplified model will be part of a probabilistic model, which, with appropriate data (e.g. p.d.f.'s) on materials, reactor operation, etc., will give the reliability of the fuel cladding.



## 3. Section of Reactor Physics

### 3.0 Introduction

Work in the section of reactor physics essentially concerns that part of the reactor behaviour that is directly coupled to the interaction of reactor materials with neutrons, e.g. reactivity, power distribution, and burn-up.

Most of the work concerned light-water reactors (LWR), but a more general study of the thorium cycle was also applied to heavy-water reactors of the Canadian CANDU-type.

Work may be roughly subdivided into the following categories:

1. Cross section processing and fuel element models,
2. Thorium cycle investigations,
3. Reactor operation studies,
4. Methods for solving the 2- and 3-dimensional neutron diffusion equation,
5. Absorber management,
6. Fuel economy and management,
7. Service.

One staff member participated for one and a half months in the containment exercise mentioned elsewhere, and one member took virtually full-time part in a comparative power cost study of different types of electricity generating system.

### 3.1 Cross Section Processing and Fuel Element Models

Programs for cross section processing and fuel element models are so intimately interlinked that the testing of these programs, which was the main activity this year, must be done simultaneously on both types of program.

Testing was done by calculation on international benchmark examples, which gives access to experimental results and calculations performed by other centres.

Studies on the plutonium benchmark from the 1973 Scandinavian Reactor Physics Meeting revealed considerable differences in details between the UKNDL 1968 and 1973 cross sections. These differences are hard to interprete without further benchmarks. Such are under way through the plutonium recycling benchmark within the framework of the Comité Consultatif en Matière de Gestion de Programme Recyclage du Plutonium of EURATOM.

A benchmark example aiming at the testing of the transport methods used in fuel boxes was formulated within the framework of the NEA Reactor Physics Committee. Solutions using our standard methods were submitted, but as yet the results have not been finally edited, mainly because too few answers have been submitted by other centres so far.

A solution to a simple example within the same benchmark by collision probability methods seems to indicate, in agreement with preliminary British results, that good, virtually exact results, can only be obtained by considerably increasing the subdivision of regions. This will be possible with the type of program based on response functions that is under development.

## 3.2 Thorium cycle Investigations

A revised set of cross sections for the isotopes $^{232}Th$, $^{233}Pa$, $^{233}U$ and $^{234}U$ was introduced into our 76-group library late last year, and an assessment was made of the possibility of using thorium in a CANDU reactor optimized for operation with natural uranium. The results were very much on line with Canadian calculations (backed up by an unknown number of experiments).

Special attention was given to the problems connected with the transition from U-operation to Th-operation. If no highly enriched uranium is available, the transition period may be quite lengthy and costly.

Calculations of Th-utilization in PWRs were also made. It appeared that the savings in uranium consumption were modest, because $^{235}U$ must be continuously added to the Th in significant quantities.

Some of these results are displayed in fig. 3.1. It shows the amount of energy, that can be extracted from 1 kg of natural uranium or thorium when used in a LWR, a CANDU or a FBR. The abscissa x indicates the fraction of fuel consumed during its use, by fissioning, by transformation into useless, heavy isotopes, or by reprocessing losses. For non-breeder reactors, it is mainly the second mechanism that lowers the values from the theoretical maximum indicated by the straight line. For breeder cycles, reprocessing losses (particularly for CANDU) are dominant. The inserted linear



scale figure reveals particularly clearly how poorly resources are utilized by present-day reactors.

## 3.3 Reactor Operation Studies

Based on data obtained from a reactor vendor, a reactor physics calculation project was started. The main objective of this project is to gain more experience in three dimensional calculations of power distribution in operating reactors.

Cross sections for overall calculations were generated and calculations on some three-dimensional critical configurations were performed. In order to handle the latter problem, some modifications of the three-dimensional nodal theory program were needed to increase the number of nodes that it could treat.

A staff member participated in the commissioning of the Barsebäck I power reactor during a period of 5 months. Increased insight was gained into the reactor physics requirements with respect to the relationship between calculation and both measurements and instrumentation during the start-up procedure, as well as during normal operation.

## 3.4 Methods for Solution of Two- and Three-dimensional Neutron Diffusion Equations and Power Distribution

The methods developed aim at an exact numerical solution of the diffusion theory equation for neutrons in few energy groups.

Finite element programs for solution in two and three dimensions were developed.

The elements are triangular or rectangular in the 2-dimensional case, and box shaped in the 3-dimensional program. At the element boundaries, continuity is required of the flux only, and not of any of its derivatives (Lagrange interpolation). The order of the calculation (i.e. the degree of the approximating polynomials) is input specified.

A number of realistic reactor calculations were carried out to establish the relation between the accuracy required and the time used by programs using the finite element method (abbr. FEM), compared with programs using finite difference technique (abbr. FDT).

As an example, we may take the 2D IAEA benchmark problem. A finite element program FEMB with rectangular elements is compared with a corner meshpoint FDT-program TVEDIM and a centre meshpoint FDT-program TWODIM. Average power densities are calculated for 20 x 20 cm meshes. Table 1 gives maximal error relative to maximal power density in per cent together with the running-time for each calculation.

Table 1

| Program | FEMB | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Order | 1 | | | 2 | | | 3 | | 4 |
| Mesh-numbers | 9x9 | 18x18 | 36x36 | 9x9 | 18x18 | 36x36 | 9x9 | 18x18 | 9x9 |
| Error (%) | 90 | 15 | 3.9 | 8.3 | 1.5 | 0.20 | 1.8 | 0.17 | 0.20 |
| Time (sec) | 400 | 600 | 4000 | 500 | 1800 | 7500 | 1600 | 5200 | 4800 |

| Program | TVEDIM | | | | TWODIM | | | |
|---|---|---|---|---|---|---|---|---|
| Mesh-numbers | 17x17 | 34x34 | 68x68 | 136x136 | 17x17 | 34x34 | 68x68 | 136x136 |
| Error (%) | 48 | 12 | 2.8 | 0.8 | 11 | 7 | 2.6 | 0.8 |
| Time (sec) | 120 | 500 | 1600 | 7500 | 50 | 200 | 1250 | 15000 |

It is seen that FEMB is slow compared with TVEDIM and TWODIM for first order calculations, but becomes much faster for high order calculations, in particular for small spacings; i.e. for accurate calculations, FEM is much to be preferred to FDT.

Another advantage of FEM is that the accuracy is less dependent on the regularity of the grid than is the case for FDT.

A fairly accurate calculation for the original 3D IAEA benchmark problem was also carried out.

## 3.5 Absorber Management

Light-water reactor operation is based on periodical refuelling. During the cycle fuel burn-up causes a reduction in reactivity, which can be counteracted by a gradual depletion of a burnable poison, e.g. gadolinium.



This depletion affects the power shape and discharge burn-up; and there is a conflict between requirements for a minimum power-peaking factor and a maximum discharge burn-up.

The problem is then to find the optimal way of distributing the burnable poison, i.e. the distribution giving minimum power production costs, all costs, including fuel-cycle and capital costs, are considered.

A study was carried out on a two-dimensional model for a PWR power reactor. The reactor core was divided into two poison control regions and the relationship between the power peaking and burn-up was investigated.

The results are illustrated in fig.3.2 where sub-index 1 refers to the inner region and 2 to the outer region. The different curves represent burn-up histories, for which a given maximal power peaking factor, f, appears during burn-up. The curve for f = 1.35 corresponds to the distribution of poison according to the Haling principle, for which f is lowest possible and incidentally constant during burn-up. For larger f values, two different solutions are possible as indicated for f = 1.57.

The maximum gain in discharge burn-up compared to a homogeneously controlled reactor is 6.15%, whereas the power peaking can be lowered by 30%, indicating a greater potential for cost reduction by a reduction in the power peaking factor.

## 3.6 Fuel Economy and Management

The program OPTI, which calculates the fuel economy and performs an optimization over several fuel cycles, has undergone several changes, some being of a purely editorial nature concerning the input/output structure of the econimical data and others being of a more fundamental nature. An improved correlation was developed to calculate the power sharing of different fuel elements as function of a few parameters characterizing the state of the fuel. A new linear optimization routine was implemented assigning a new kind of restraint to the number of fuel elements of a particular type, as characterized by state variables, that are allowed to be loaded into one zone in the reactor. The restraint only allows this number to be either zero or above a specified minimum. This eliminates fuel batches that are too small to be handled practically.

3.7 Service

Follow-up calculations were carried out on four Danish fuel elements irradiated in the Kahl power reactor. For this purpose the CDB 3 code was slightly modified allowing restarts with altered conditions (i.e. void, power, cross sections) resulting from burn-up and changes in the mode of operation.

Calculations of fast flux (essentially the flux above 1 MeV) were carried out for several rigs in the DR 3 reactor in order to predict radiation damage.

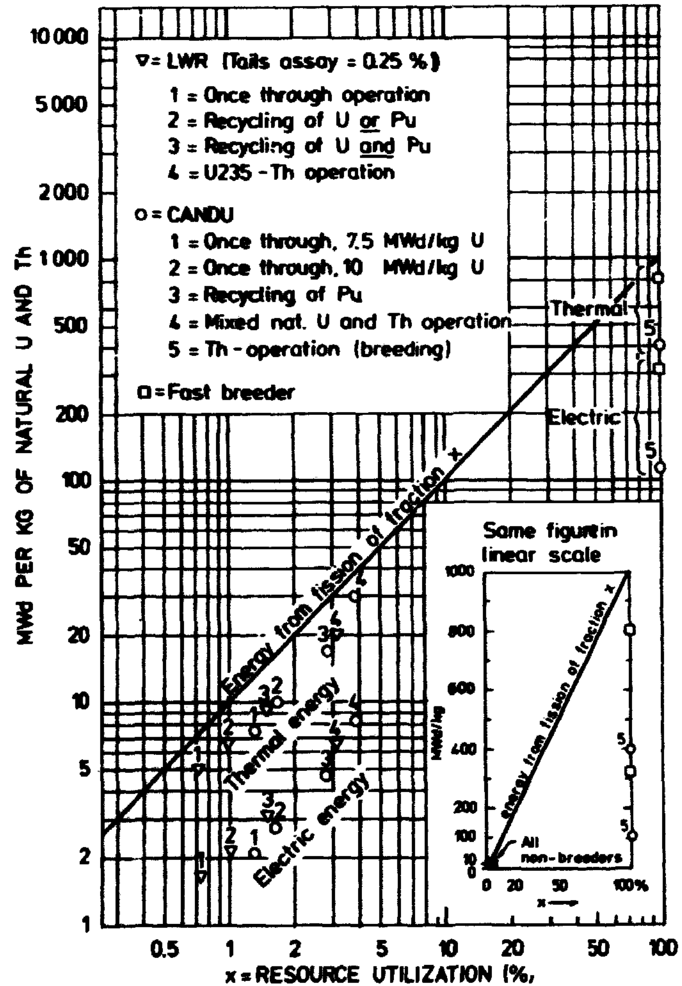
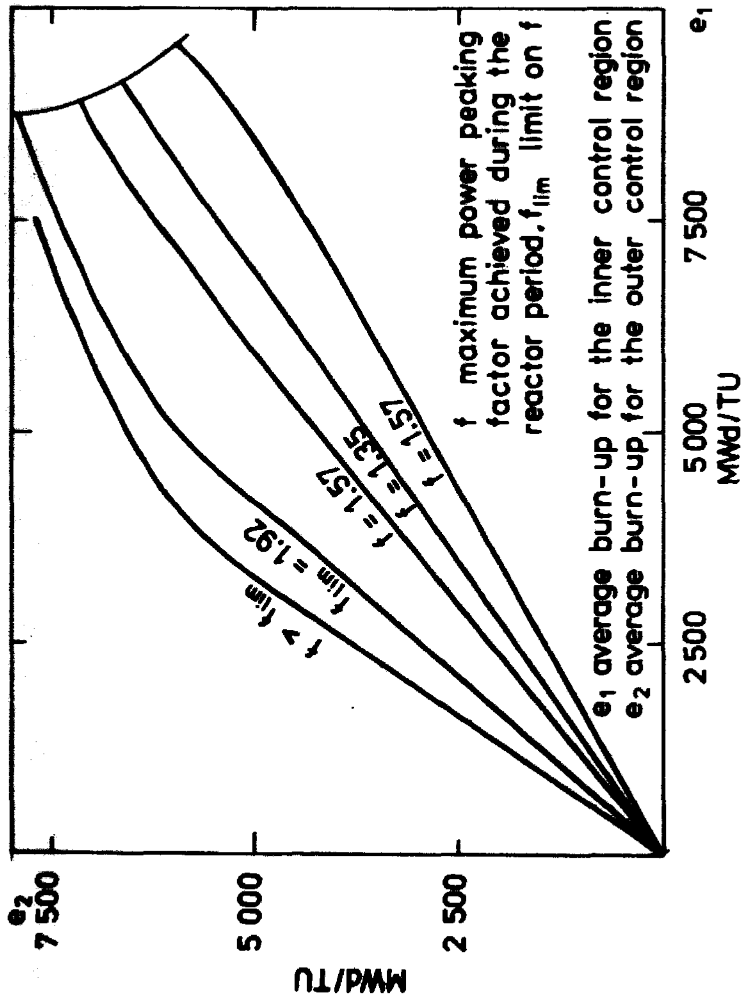



Fig. 3.1. Thermal and electric energy produced per kg of natural uranium and thorium for various reactor types and various types of operation.

(Yearly consumption of electric energy in Denmark ~600.000 MWd).

Fig. 3.2. Terminal state for burn-up [illegible]






## 4. Section of Heat Transfer and Hydraulics

### 4.0 Introduction

The objectives of the section are to gather know-how and to develop models for thermodynamic and thermohydraulic phenomena in connection with the use of nuclear energy.

The main working areas were:

1. Steady state thermohydraulics
2. Accident analysis
3. Containment analysis
4. Participation in international accident experiments

### 4.1 Steady State Thermohydraulics

Work is directed towards the development of physical models and computer programs for the description of heat transfer and flow-related processes in nuclear power plants. Main efforts concern the prediction of core void distribution and the determination of burn-out limits.

#### A. The Scandinavian Subchannel Analysis Project

The SDS-project was terminated last year. The object of the project was to develop a computer program for the calculation of the three-dimensional temperature-, flow- and void-distributions plus burn-out limits for BWR and PWR fuel elements. The code development work was backed up by experiments in 7-rod, 9-rod and annulus geometries.

This year, efforts were directed towards the verification of the program by comparison with available experimental data.

A comparison of the SDS-program against data from the Swedish 9-rod experiment has been reported in Sweden (SDS-report No. 85). It is concluded that the agreement between calculational and experimental results is good. On the basis of comparative calculations, this report also concludes that the SDS-program yields better results than the COBRA-II program.

Another comparison was made between the SDS-predictions and experimental data obtained by General Electric in a 9-rod bundle. This work was recently completed and the report will appear early in 1976.

The experimental data considered in this study cover

- Measurements of isokinetic, exit mass flux and enthalpy distributions under single- and two-phase conditions.
- Measurements of axial pressure drop under single- and two-phase conditions.

It is concluded that

- The calculated single-phase mass flux distribution agrees well with the data.
- The program is unable to predict the trend in the exit steam quality distribution. (The experimental corner sub-channel quality is below the average, although the power ratio is above).
- The Moody single-phase friction factor, corresponding to a relative roughness of $10^{-4}$, is too small. A relative roughness of $10^{-3}$ yields better agreement between calculations and experiment.

A comparison between the SDS and COBRA programs based on the General Electric data shows that

- The SDS single-phase mass flux distribution results are in best agreement with the data.
- No significant difference could be detected with respect to the accuracy of the two-phase calculations.

B. Film Flow

The study of the annular flow regime characteristic of high-quality two-phase flow was continued, and has now become the subject for a doctoral dissertation by a postgraduate student from the Technical University of Denmark.

A new film-flow model based on a "microscopic" description of the relevant physical phenomena was developed. It is capable of predicting film-flow rates for tubular and annular geometries both under fully developed adiabatic conditions and under diabatic conditions leading to "burn-out" i.e. dry-out of the film. The theory was presented in the paper "Film Flow in Annular Geometries" pre-



sented at the European Two-phase Flow Group Meeting in Haifa in May 1975.

One of the largely unresolved theoretical difficulties concerns the description of the droplet deposition process. This is particularly true of annular geometries where it appears that the assumption of droplet diffusion alone is incapable of explaining the observed higher deposition rate on the tube compared to the rod. An attempt is being made to solve the problem by considering the droplet-gas interaction in detail. The experiments, which are planned to take place in the SEHT high-pressure water loop, are designed to guide theoretical efforts. (See section 5 of this report).

4.2 Accident Analysis

An accidental breach of the primary coolant system of a water reactor is called a loss of coolant accident. The evaluation of the consequences demands the study of the related thermohydraulic phenomena, a task which calls for general theoretical models (computer programs) backed up by relevant experimental results.

Most of the Section's work in this field is done within the framework of the "NORHAV project" in co-operation with Finland, Norway, and Sweden. Danish and Norwegian contributions are mainly development work on new blow-down and core heat-up computer programs, while Finish and Swedish contributions consist of updating American computer codes. Sweden further contributes by performing core cooling experiments in a 64-rod cluster.

A. Blow-down

Our primary efforts in this area were directed towards the further development of TINA, the dynamic subchannel code. TINA can perform a thermohydraulic analysis of BWR and PWR cores during blow-down. The program solves the conservation equations for the following properties:

- the mass of the steam and water phase individually
- the energy for the two-phase mixture
- the mixture axial and lateral momentum.

The steam is assumed to remain saturated while the water may be subcooled or superheated. A "flashing function" specifies the rate of evaporation or condensation (NORHAV-D-012).

A method for calculating the axial slip velocity was developed. The slip velocity is expressed as a function of the fluid acceleration and of the flow regime (NORHAV-D-015).

The coupling of the thermal behaviour of the fuel to the hydraulics was accomplished by solving the (radial) heat conduction problem with the appropriate boundary conditions for each subchannel and each axial mesh point. The heat conduction problem is solved by the subroutine RODTEMP by an implicit numerical method (NORHAV-D-011). Deformation of the cladding (ballooning and rupture) cannot be predicted at present, but TINA has the capability of handling variable flow areas.

Fuel rod behaviour during loss-of-coolant accidents has been and is currently the subject of Masters theses for students from The Technical University of Denmark using FRAP-T as the point of departure. However, a suitable model that might be used in TINA has not yet been developed.

A point kinetic model based on a stable implicit numerical scheme was programmed and tested. This model has not yet been implemented in TINA.

The numerical solution that was developed for TINA is of a linearized implicit type. It has no serious instability problems even at large time-steps. It is capable of handling flow reversal with opposite flow directions in neighbouring subchannels.

TINA was tested against the experimental data for steady-state, single-phase flow in a partially blocked subchannel geometry obtained by D.S. Rowe et al. Preparations are presently being made to test TINA against data from the LOFT semiscale experiments.

Parallel with the development of the subchannel code, TINA a one-dimensional version, TPL3 (a newer version of TPD3) was used as a test-bed for some of the physical ideas that are and will be incorporated in TINA. TPL3 was, for example, used to test various flashing functions and ideas for calculating the slip velocity. Computations of critical flow rates performed by TPL3 were reported in NORHAV-D-015.

TPD3 was used in our participation in the ad hoc group CSNI



standard problem program. The results for the USAEC Standard Problem One for Blowdown (the Edwards and O'Brien single pipe blowdown) are presented in Risø-M-1785.

TPL3 is capable of handling closed loops and will also be used for testing of our new "pump model". This model is based on the observation that the two-phase performance of a pump (head and torque) can be predicted from the single-phase pump characteristic if a homogeneous flow with a suitable density (void) is assumed. Correlations for the appropriate internal mean void were established on the basis of experimental data for the LOFT semiscale pump. It was found that only under extreme conditions (e.g. reverse flow with normal direction of rotation) does the internal mean void deviate appreciably from the simple average between inlet and outlet voids.

## B. Core Heat-up and Emergency Core Cooling

During the year the computer code REMI/HEAT-COOL (RHC) was improved in several ways. These cover two areas, improvements in the physical models and improvements in the numerical methods.

In many cases of practical interest, the simple one-dimensional rewetting model, the Yamanouchi model, with heat conduction only in the axial direction in the cladding, fails to give correct predictions of the rewetting time. Many authors have shown, however, that the rewetting phenomenon is essentially a two-dimensional problem, i.e. the radial heat conduction in the cladding must also be taken into account. This is illustrated in figure 4.1. A two-dimensional model was therefore developed and correlated with published experimental results. The model is now incorporated in RHC.

A significant amount, 10 - 30%, of the heat transfer from the fuel to the coolant is due to thermal radiation to both droplets and steam. Only absorption in the droplets was taken into account in the old version of RHC, but it has been shown that absorption in the steam is, in fact, just as important as absorption in the droplets. A model for thermal radiation to both steam and droplets was therefore developed and incorporated in RHC.

Reflection of the thermal radiation from the rod surfaces was assumed to take place isotropically, but this may lead to an underprediction of maximum surface temperatures of 30 - 50°C. The rod surfaces should be divided into several sub-surfaces in order to

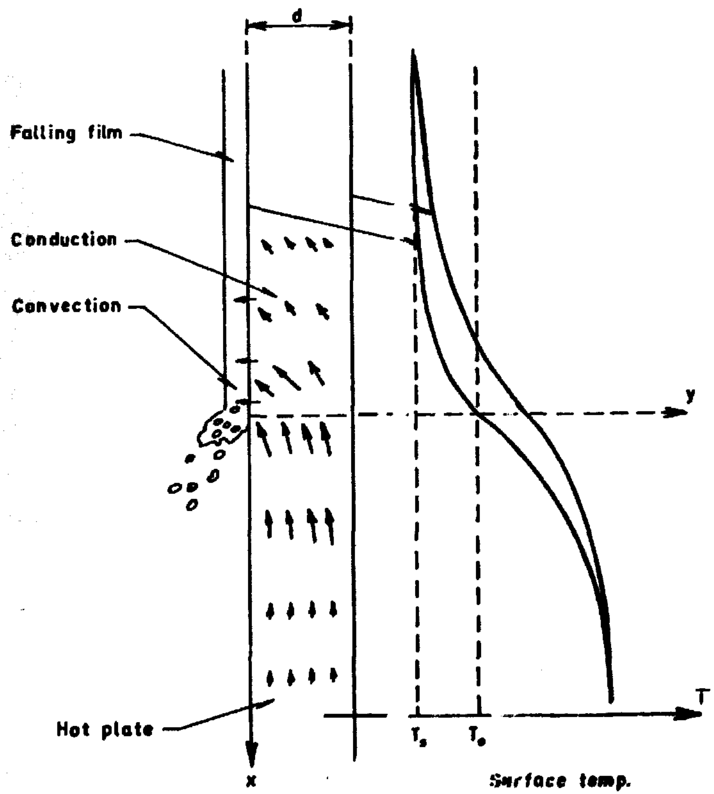



Fig. 4.1 Physical Phenomena in Rewetting



treat the reflection problem correctly. This will, however, increase the necessary computer time to a prohibitive degree. Instead, a transport-corrected thermal radiation model was developed and correlated with more exact calculations and experimental results. The model is now incorporated in RHC.

The presence of droplets in a steam atmosphere has two effects on the convective wall to steam heat transfer. The bulk steam temperature will be lowered towards the saturation temperature, increasing the heat transfer. Secondly, the temperature profile of the steam close to the wall will acnieve a steeper gradient, enhancing the heat transfer. A model accounting for these effects was included in RHC.

The steam temperature may vary considerably across the fuel bundle, from saturation close to a wetted surface to a temperature in the order of 500°C superheat in the central region. The use of an average steam temperature will lead to errors in the local heat transfer rates. A model dividing the bundle into a central region and an outer region with different steam temperatures was developed and included in RHC (figure 4.2).

The flow of spray water to the bundle may be reduced in the case of large spray flow rates or increased steam flow due to counter current flow limitation, CCFL, resulting in reduced efficiency of the emergency core cooling. A model for CCFL was developed and incorporated in RHC.

Finally, the pressure in the drywell may change during the LOCA and a model allowing the use of a prescribed pressure transient was therefore included in RHC.

The new version of RHC was successfully tested against a few experiments from General Electric, USA, and from AB Atomenergi, Sweden. The verification of the RHC will continue in the coming year with results from several experiments placed at our disposal under an agreement between General Electric, USA, and Risø. Moreover, verification of RHC will be carried out with results from the Swedish GÖTA-loop under the NORHAV-collaboration.

Development of a reflooding model was initiated in collaboration with General Electric. The model will be based on RHC and the GE-developed program REFLOOD, which has a detailed modelling of the primary system. A member of the section is currently attached to

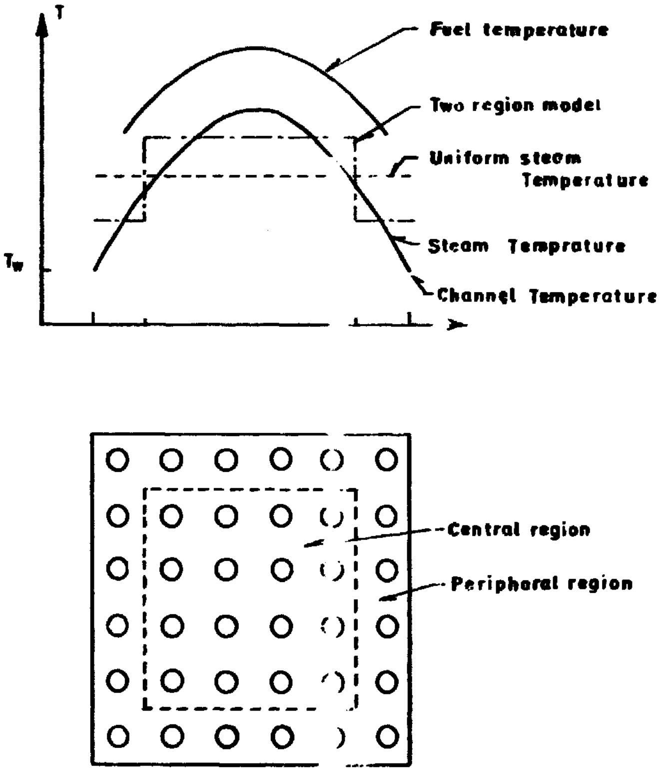



Fig. 4.2 Two Region Steam Temperature Model



General Electric, San Jose, USA, to take part in the development of the reflooding model.

The problems encountered in the annular downcomer region of a PWR reactor during the ECC-phase were the subject of a Master's thesis for a student from The Technical University of Denmark.

## 4.3 Containment Analysis

Work on containment analysis derives largely from the experience gained from participation in the Marviken containment experiments. A selected set of these experiments was analyzed by our containment program CONTAC-II. This work is reported in SäH-2-75. Much attention has been paid to the pressure oscillations discovered in the Marviken experiments and elsewhere cf. also the TECPO-project, (see p. 31). In view of this, the integration method in CONTAC-II had to be revised because of the possibility of numerical instabilities. A new solution method for the system of first-order differential equations was incorporated in the routine. A one-dimensional transient heat conduction program with accuracy specification was developed. Furthermore, a new vent flow model has been incorporated. Special reports on these matters are under preparation.

The modified version, CONTAC-III, was also applied with satisfactory results to the above-mentioned Marviken experiments. The new version of the code exhibits no numerical oscillations.

A comparison of the CONTAC-III program with the American reference program, CONTEMPT-LT, has been reported (SHH-7-75). A comparison between CONTAC-II and CONTEMPT-PS, on the basis of the Marviken data was made in collaboration with CEN, Saclay.

A project in connection with a doctoral dissertation and aiming at the study of the containment pressure and temperature response during loss of coolant accidents (LOCA) was initiated in the spring of 1975.

The containment is analyzed by representing rooms and connections by coupled "pipes". The corresponding set of coupled, nonlinear differential equations are separate equations of continuity and energy for the gas and liquid phases and a mixture momentum equation. The model takes into account that the gas and liquid phases may have unequal velocities, and that both may be in a state of non-equilibrium.

The following propertie. of this model should also be mentioned:

1. Water level in pipes can be detected and followed (for pressure suppression (PS) containment simulation).

2. Heat exchange between the phases and between the phases and the surroundings is simulated by simple heat exchange models.

3. Heat conduction in walls is simulated by the one-dimensional, unsteady Fourier equation for heat conduction in a slab geometry.

The model will be verified using data from the Marviken containment experiment.

4.4 Participation in Experiments Abroad

The first Marviken project, MXI, was terminated at the beginning of the year. It was the first project in the international cooperation on full-scale containment experiments conducted in the pressure suppression (PS) containment of the abandoned nuclear power station at Marviken, Sweden. The experiments were carried out from August 1972 to May 1973, while the analysis and the reporting (comprising fifty-nine reports) continued until early this year.

The experiments included studies of

- containment pressure and temperature response to a loss of coolant accident,
- fission product (I + Xe) transport in the containment and leakage through the walls under accident conditions,
- various component tests under accident conditions.

Emphasis was laid on studying the phenomena relating to the pressure and temperature in the containment during blowdown from the reactor vessel.

The test programme comprising 16 blowdowns included an investigation of the effects of varying a number of parameters, such as simulated break size and type (steam or water), location of break, vent pipe submergence depth and flow area, and initial pool temperature.

A large amount of information that can be used for the verification of containment computer codes is available from the project. A limited amount of program verification was performed within the project. It was found that due to well-known inherent conservative features, the available codes generally overpredict the containment pressures.



The sixteen blowdown tests demonstrated in full-scale the applicability of the pressure suppression concept as an effective means of condensing large amounts of steam released during an accident, and thus of controlling the containment pressurization as intended.

During the experiments pressure oscillations of significant amplitudes (although not to an extent that violated the integrity of the containment) were discovered in the containment, particularly in the vent system and in the suppression pool. The instrumentation was not suitable for detailed analysis of this phenomenon and, as the oscillations are not fully understood, further investigations were recommended by the project.

The second Marviken project, MXII-CRT, (Containment Response Tests), was initiated as a natural consequence of the above recommendation. The project, which started early in 1975, is a multinational collaboration with participation from the German Federal Republic, France, Holland, the United States of America, Japan, and the four Scandinavian countries. The budget amounts to about 15 million SW.kr.

The primary objective of the project is to improve the understanding of the mechanism behind, and identify the significant parameters governing the amplitudes and frequencies of pressure oscillations like those previously observed in the blowdown experiments of Marviken I. A further objective is to investigate other more general containment response phenomena. A related objective is the characterization of the structural response to these pressure oscillations.

The test programme settled for these purposes comprises eight blowdown runs with variation of the parameters thought to influence the pressure oscillations. These parameters include vent pipe submergence depth and flow area, pool surface area and mass, pool temperature, vent flow path geometry, vent flow rate and composition of flow.

The project is still at its planning and preparatory stage.

Much work has been done on the mechanical equipment thus improving the experimental conditions. A comprehensive instrumentation and data acquisition system is being established. It is intended to retain (with minor modifications) the digital sampling technique used during MXI for quasi-steady measurements, whereas pulse code modulation techniques will be implemented for dynamic measurements. The programs and plans for the data conversion and evaluation are under preparation.

Experiments are planned to start early in 1976 with an expected duration of about eight months, followed by an editorial phase of six months. Thus the planned termination of the project is the winter or spring of 1977.

TECPO (Theoretical Efforts on Containment Pressure Oscillations) is an inter Nordic project associated with the MXII-CRT project described above. Via joint theoretical and experimental efforts, it is intended to create a theoretical background for the project, and participation is open for other MXII-members, but none of them have joined so far.

For the purpose of the experimental investigations, a small-scale PS-containment model was built and operated for sixteen blowdown runs, constituting approximately half the planned experimental series. Such a small scale facility offers a much higher degree of flexibility than the large scale containment. The dependency of the oscillations (which generally have higher frequencies than in the large containment) on various geometrical configurations, as well as on pool temperature and changes in flow rate and composition, has already been investigated experimentally. However, much data from these blowdowns still remains to be evaluated.

Theoretical efforts have concentrated so far on literature searching and on the prediction of eigenfrequencies for the containment system treated as a system of coupled pipes with standing waves. This quite simple model is used because the objective of the project is not to make large computer programs, but rather to clarify the phenomena involved in the generation of the pressure oscillations. The work will not rely only on the hypothesis that the oscillations are due to an instability in the processes governing the condensation of steam in the cold pool.



## 5. Section of Experimental Heat Transfer (SEHT)

Work mainly concerns experimental research on heat transfer problems in nuclear power plants.

The experimental work is carried out in close co-operation with the Section of Heat Transfer and Hydraulics in order to support the theoretical work of the department. The new facility for carrying out such experiments, the high-pressure water loop, is installed in the experimental hall.

### 5.1 High-Pressure Water Loop

The pressure test of the loop took place in January 1975. The pump performance test, with both cold and hot water, was successfully carried out early in February 1975 in the presence of two engineers from the Japanese manufacturer. The pump fulfils all the specifications.

The instrumentation, control, and safety system was installed by the SEHT staff supervised by the Electronics Department.

The equipment was finally approved by the authorities in July 1975 except for the safety valve, which had to be replaced by a more efficient type within 6 months time. Exemption was given for making test runs with the original safety valve.

During the test runs in the autumn severe vibrations occurred in the main flow-regulating valve at flows below 3 kg/s. It was necessary to replace the plug and seat in the valve. The spare parts were constructed and fabricated by SEHT because of an unacceptably long delivery time for original spare parts. The valve now seems to operate satisfactorily.

## 5.2 Test sections

The annulus test section, the regenerators and the heat exhangers, which were earlier used for experiments in Stockholm (SDS-project), have been modified for the SEHT high-pressure water loop.

Due to an inexpedient design of the preheater, it was impossible in the Stockholm experiments to carry out adiabatic tests at higher steam qualities than 20 %, and a new preheater was therefore constructed and manufactured. It consists of nine tubes (9.5ø/8.2ø) connected in parallel and this new design will fulfil the requirement for high outlet steam qualities.

All the above equipment was pressure-tested and fitted into the loop, and introductory tests have just been started.

After completion of these introductory tests, the annulus test section is to be replaced by a tube test section currently under construction. This test section has been designed with tube couplings making it possible to replace the tube by a tube of other dimensions. It has been decided to carry out measurements using this test section with tube diameters of 5, 10 and 20ømm.

The experimental programme for the annulus test section and the tube test section will include the following measurements:

1. Determination of burn-out,
2. Film flow,
3. Thickness of the water film,
4. Pressure drop.

Apart from the burn-out, these measurements have to be carried out under both adiabatic and diabatic condition.

## 5.3 Test of nozzles

In connection with the 64-rod emergency spray-cooling experiment in Sweden, SEHT examines different nozzles.

The measurements determine the distribution of the water in different subchannels of the element, as well as the sizes of the droplets, as a function of the pressure drop over the nozzles.



By use of a very short exposure time (1 μsec), it is possible to quick-freeze the drops on a photograph.

Measurements carried out on the photographs give the diameters of the droplets.

## 6. Section of Dynamics

### 6.0 Introduction

The aim of the work in this section is to develop and maintain dynamic models of power plants, reactors and associated control systems, and to study the dynamic behaviour of the systems during both normal and abnormal working conditions. The results can be used for evaluating the safety of power plants and for investigating advanced control and protection systems.

In the past year work concentrated on the following subjects:

1. A one-dimensional reactor model for xenon calculations.
2. A one-dimensional model of a PWR power plant.
3. A one-dimensional model of a BWR power plant.
4. Study of rod ejection transients in a BWR.

In addition, some work was carried out in order to improve the numeric routines in the 3-dimensional BWR model mentioned in the last annual report.

### 6.1 Xenon Oscillations

Nuclear power plants used in a load-following mode will excite xenon transients which may force physical quantities, such as heat flux and mechanical stresses in the fuel, beyond the safety limits if the local powers are not properly limited. This presents a problem especially in large reactor cores, which may be regarded as a collection of loosely coupled smaller cores. It is well known that some types of reactor e.g. the BWR, are more stable with respect to xenon oscillations than other types, e.g. the PWR, because of the large negative void coefficient of the former compared with the latter.

One of the main reasons for considering the problem is that the quantities responsible for the oscillations, namely the xenon and iodine concentrations, are not amenable to direct measurement; and, as the time constants involved are rather long, ~5 hours, the operator has little possibility to manipulate his controls correctly in order to damp the oscillations.

One way to control the xenon oscillations is to calculate the time-dependent xenon and iodine concentrations by means of a reactor simulator that gives a correct picture of the conditions inside the



power plant. The output of the simulator, power distribution and other pertinent quantities, are compared with measured values and the differences are used to feed a process controller which positions control rods in the reactor in such a way that oscillations are damped.

The first step towards a full reactor simulator for xenon transients is the development of a one-dimensional core model with a fast integration procedure in order to study oscillations in a simple reactor. Furthermore, as the model is aimed at a PWR-representation, the conditions of the coolant are considered constant in the present version. The model allows the power of the reactor to be taken as a free variable, in which case it will show very large amplitudes; or it may be fixed, assuming an ideal global power control, in which case the local power will oscillate.

For illustrative purposes, two figures are given that show oscillations in a reactor core, divided into 10 sections, with a stationary power of 2 GW. A control rod is removed from the first core section during one hour and inserted again during the next hour. Figure 6.1.B shows the conditions with the power control acting on a global poison, and the oscillations in the perturbed section are clearly seen. Figure 6.1.A shows the results of the same perturbation without power control, which results in a shut-down of the reactor lasting until the xenon concentration has decreased sufficiently, whereafter the power increases rapidly, again resulting in a xenon poisoning, and so on.

### 6.2 A One-Dimensional Model of a PWR-Power Plant

The basic version of the model which works as a real time simulator was developed earlier and mentioned in the last annual report.

Some improvements giving a larger dynamic range were introduced, and a routine for calculation of transfer functions was included in the digital part of the model. The measurements on the simulator are carried out by means of a multi-frequency signal and run semi-automatically so that transfer functions are calculated quite easily. An example of a function is shown in figure 6.2. Such functions will be used for investigations of power control systems.

The model was further used for calculation of transients for abnormal occurrences, such as loss of power load, but it appeared to be less adequate for this purpose partly due to the limited dynamic range, but also due to the limitation in the number of components in the cooling system and steam circuit. For calculation of specific abnormal transients needed for safety analysis, a pure digital version is now being developed. This uses two cooling loops with improved steam generator models and a feedwater system for the steam generators. The programming is based on a simulation system, DYSYS, from Kernforschungszentrum Karlsruhe. Routines for most of the single-system components were programmed and some were coordinated as a unit for the total plant simulation. The ratio of computing time to real time will be about 35 to 1 for a Burroughs 6700 computer.

## 6.3 A one-Dimensional BWR Plant Dynamic Model

### 6.3.1 A detailed digital model

The project "A BWR Plant Dynamic Model", started almost three years ago, has now been completed, and has resulted in a computer program, "BWRPLANT". The dynamic model is based on the earlier developed steady-state model "TURBPLANT" (DYN-1-75, 1975). The model comprises a boiling water reactor, a high-pressure turbine, a combined moisture separator-reheater, a low-pressure turbine, a condenser, feedwater heaters and a feedwater pump. All parts of the model are treated one-dimensionally except for the neutron kinetic part of the reactor model, which is based on the point kinetics equation. Different plant control systems are also taken into account, e.g. the regulation valve at the inlet to the high-pressure turbine, the regulation of the bypass valve, the reactor pressure control system, the recirculation flow control system and some water level control systems.

The main purpose of the model is to study different transient occurrences during both normal and abnormal operating conditions; in addition, control systems can also be evaluated.

A disadvantage of the model is its need for very detailed descriptions of the technical data of the considered plant for use as input parameters.

Because it proved almost impossible to obtain all these data for a single plant, only some parts of the model were tested. One of these tests includes a comparison of the reactor model used in



"BWRPLANT" with the three-dimensional reactor model "ANDYCAP". Both of the models were exposed to the same steam load perturbation and the results of the two calculations were compared. The data of the Oskarshamn II reactor were used in the calculation. Before the "BWRPLANT" calculations were initiated, a series of steady-state calculations with "ANDYCAP" was made in order to determine void, Doppler and moderator temperature reactivity coefficients of Oskarshamn II to be used as input parameters for "BWRPLANT".

The results of the calculation are shown in figure 6.3.A with the steam load perturbation, the nuclear power and reactor pressure. We see very good agreement between the results of the two models, when considering their great difference. This might be because a steam load perturbation causes no local effects but only overall effects, in which case the rather simple treatment in "BWRPLANT" is sufficient.

In figures 6.3.B and 6.3.C are shown some of the results that can be obtained from the model by calculating a typical transient for a 600 MWe BWR plant. At the top of figure 6.3.B the perturbating quantity is shown as a demand for a power increase of 10% within 1 s (the dotted curve). The unbroken line shows the response of the turbine to this power increase. Below is shown the steam flow to the high-pressure turbine that is necessary to meet the power demand. In order to obtain this fast increase in steam production, the reactor pressure is allowed to decrease until the recirculation pumps have established a new core flow pattern through a change in the pump valve position. The two lowest curves on figure 6.3.B show the reactor pressure and the core inlet flow velocity. Finally, figure 6.3.C shows the corresponding effect on the thermal power and the nuclear power.

Besides the results mentioned above, the model also calculates the pressure at the inlets to the high-pressure and the low-pressure turbines, the extraction flow to the feedwater heaters, feedwater flow and feedwater temperature, reheater flow and temperature, and several other quantities.

### 6.3.2 A simplified hybrid model

A hybrid model is being developed in order to be able to calculate transients for normal operating occurrences and to study control strategies in a true time scale. For this purpose the reactor model will be the most detailed part of the model, namely a one-dimensional model, while the turbine and the reheaters will be described by some few lumped models derived on the basis of the detailed model mentioned above. The reactor model is programmed and tested. The rest of the work was postponed to next year.

## 6.4 BWR-Control Rod Ejection Accident Analysis

The purpose of this project is to estimate what consequences a hypothetical control rod ejection would have for the reactor vessel and its internal components.

The project consists of three major parts:

a) A description of the initiating event and a calculation of the control rod velocity during the ejection.

b) A calculation of the power generation, the coolant flow and the temperature distribution as a function of time.

c) An estimation of the consequences for the reactor internal structure.

For each control rod in a BWR, there is below the reactor core a tube (figure 6.4.A) serving as a guide for the withdrawn rod. The guide tube penetrates the bottom of the reactor vessel; outside the vessel it is prolonged by the control rod drive house.

The severest accident occurs if the control rod drive housing and the guide tube separate (a support structure must fail too). In such a case the pressure at the bottom of the guide tube decreases, and the coolant starts to stream down through the tube and out into the containment. Because of the huge pressure difference acting on the control rod, the rod receives an enormous acceleration out of the core. In a short time critical flow is attained, and the pressure at the bottom starts to increase. When the pressure increases the hydraulic force on the velocity limiter (figure 6.4.A) reduces the acceleration and finally the rod gets a steady motion.

A program called RODACC was written to calculate the control rod velocity and the coolant leak rate. This program is also able to handle rod drop accidents where the control rod drops out of the core under the influence of gravity alone.

As an example, results of a RODACC are shown in figure 6.4.B.



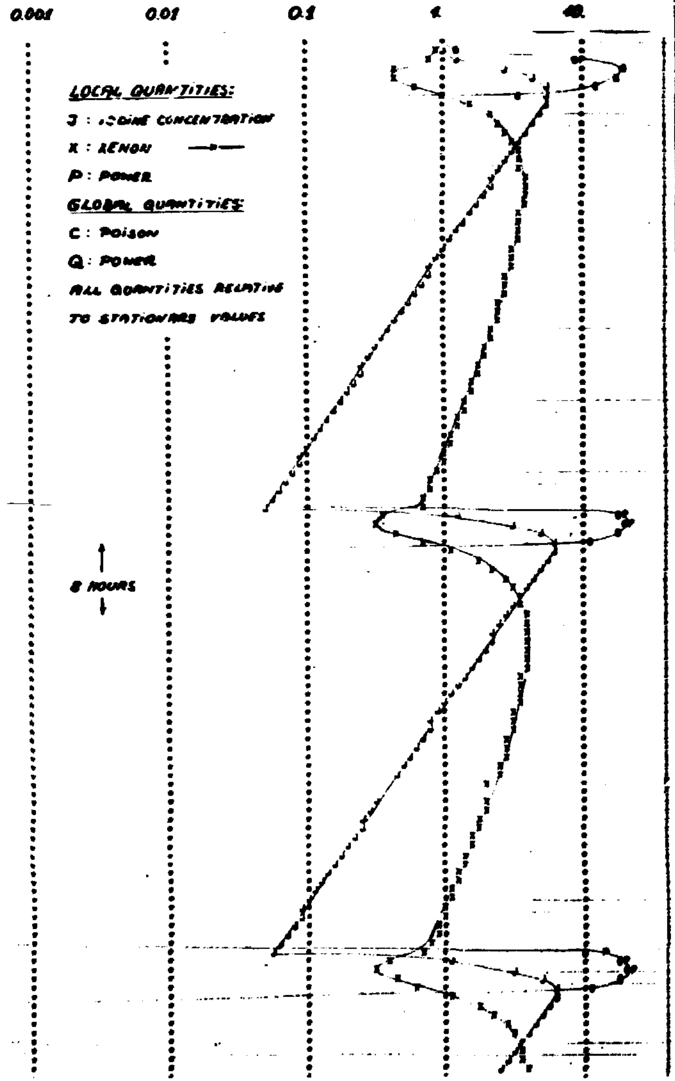
RODACC runs were used as input to a three-dimensional BWR dynamic code based on the program ANDYCAP. The original version of ANDYCAP was intended for normal reactor conditions, but it proved inadequate for low power calculations. Because the maximum reactivity of one fully inserted control rod is highest when the reactor power is low, a rod ejection accident at low reactor power is the most serious. By making the energy calculations in ANDYCAP more accurate, it is now possible to handle low power problems without difficulties.

As a test case, the control rod in a small core with only 60 fuel boxes was ejected at low reactor power. Some results from the calculation are shown in figure 6.4.C.

Future work on this project will be to make calculations on full-scale power reactors and to estimate what consequences different control rod ejection accidents might have for the power plants.

## FIG. 6.1A

XENON OSCILLATIONS






## FIG. 6.1B

XENON OSCILLATIONS

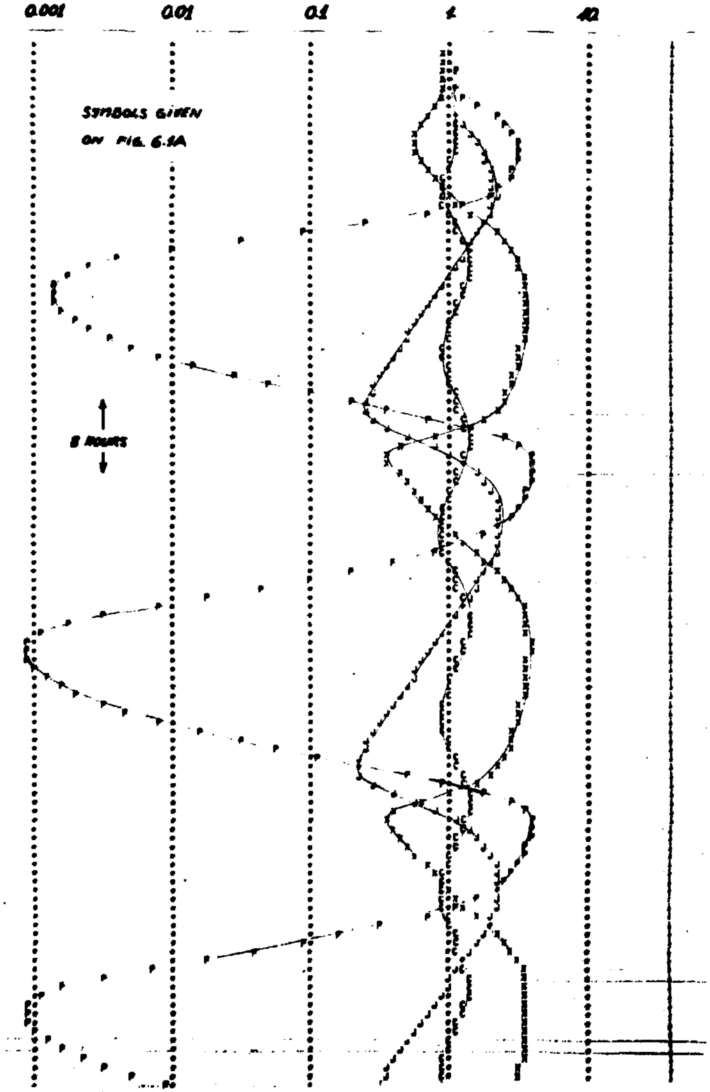
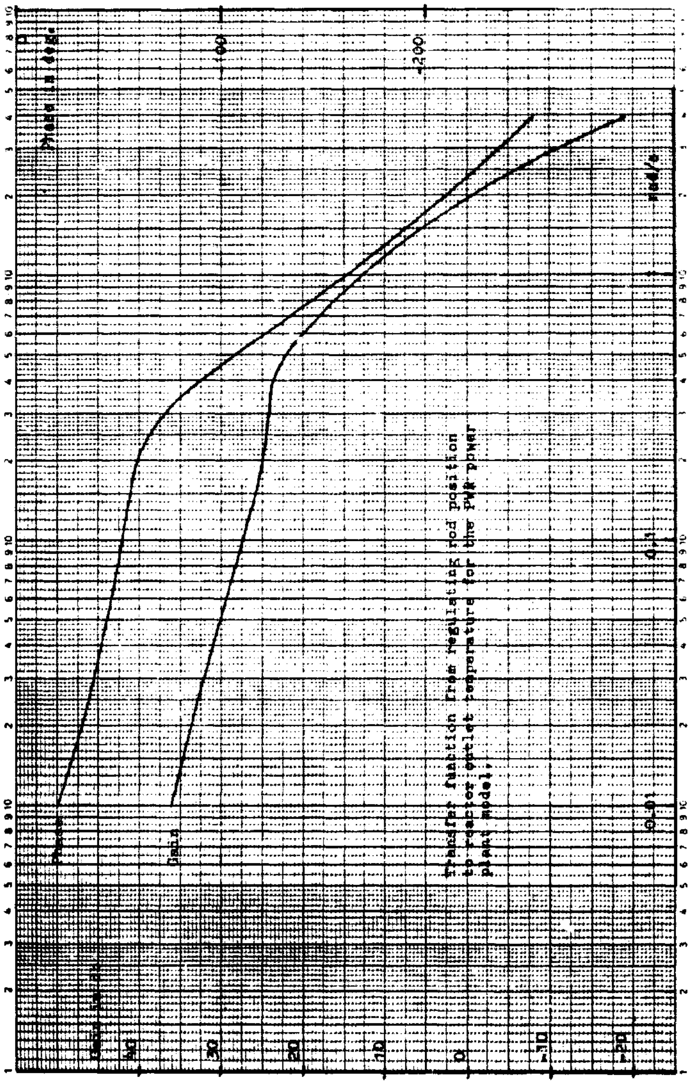


Fig. 6.?




Transfer function from regulating rod position to reactor outlet temperature for the PWR power plant model.



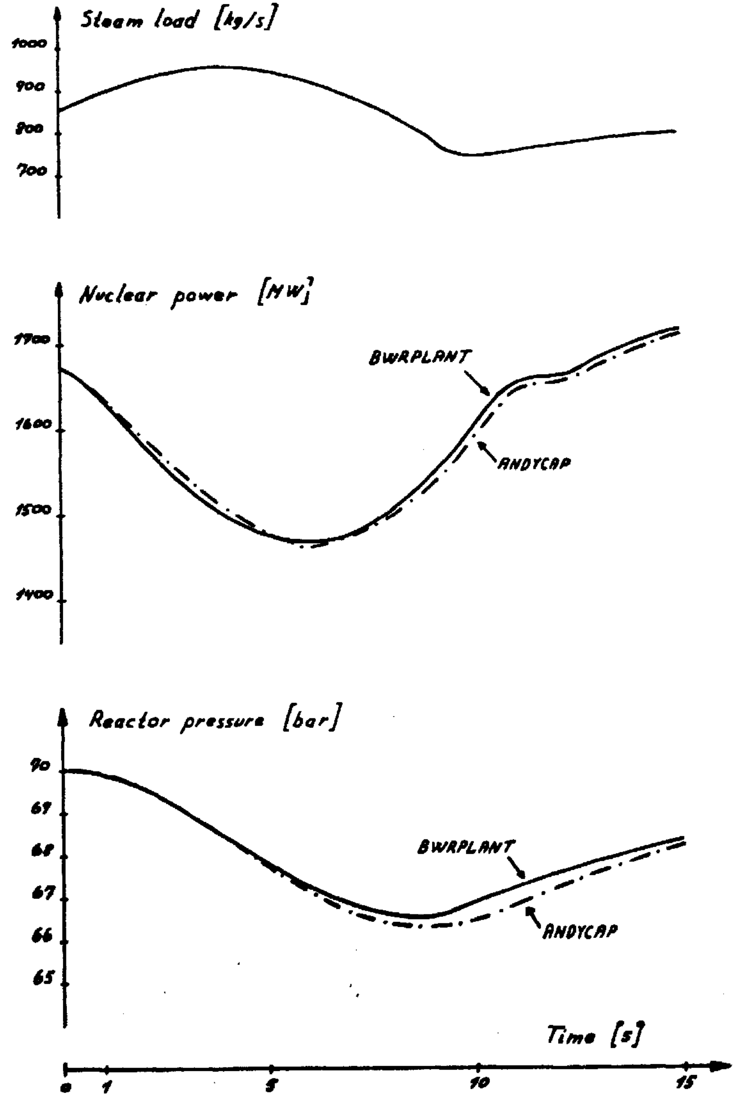



Fig. 6.3.A

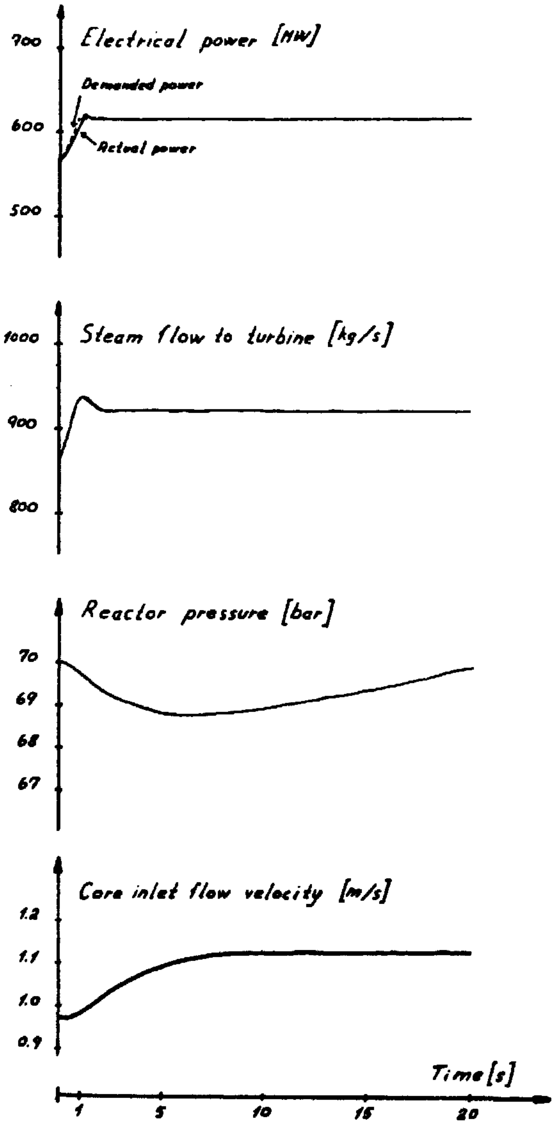



Fig. 6.3.B



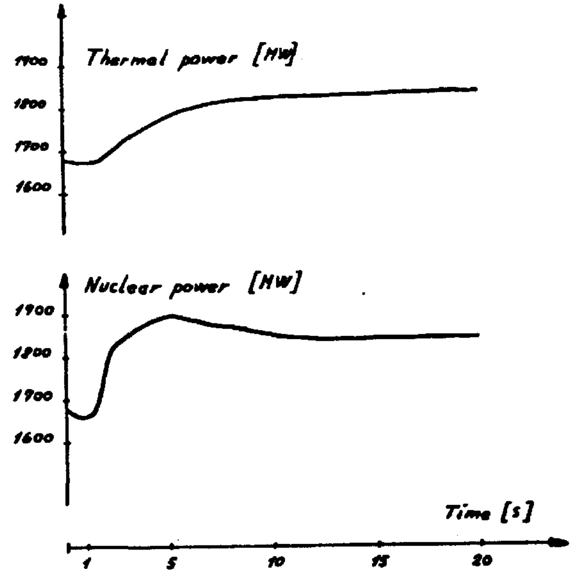



Fig. 6.3.C

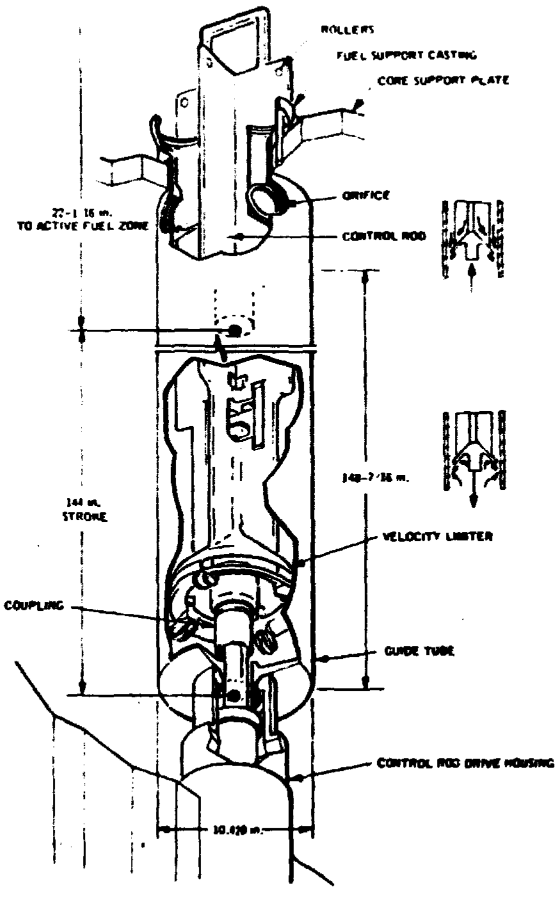



**Fig. 6.4.A Control Rod in Guide Tube**



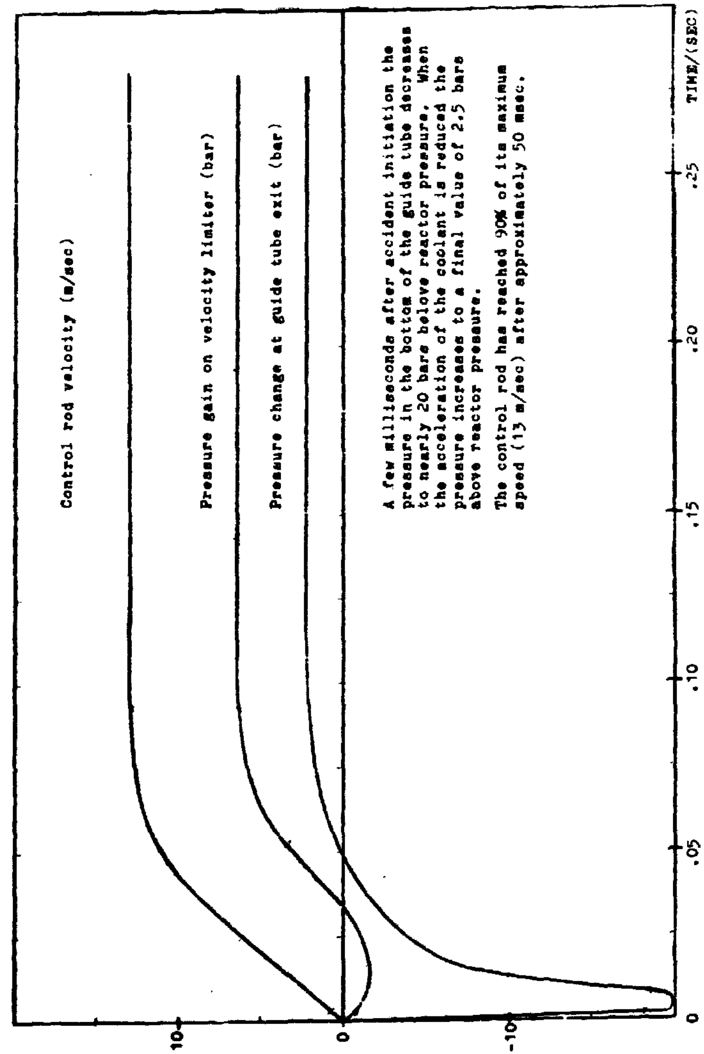
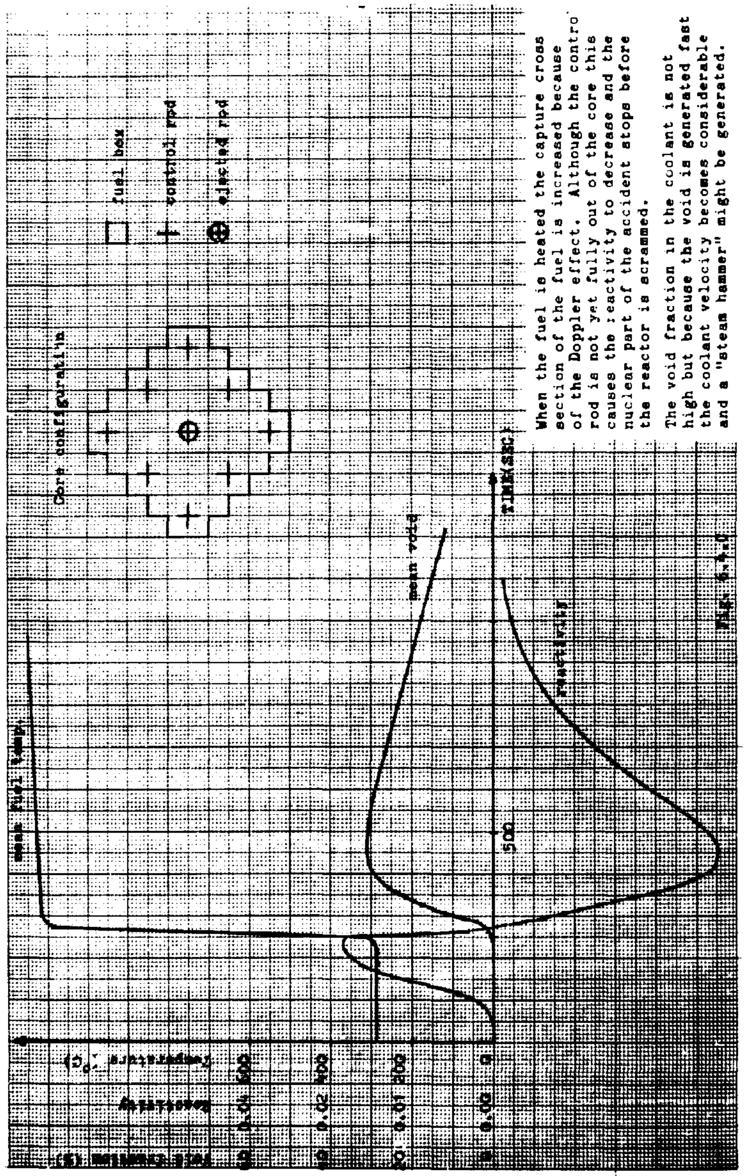



Fig. 6.4.B






# 7. The Danish Reactor No. 1

## 7.0 Introduction

The reactor was in routine use for neutron radiography and training purposes.

## 7.1 Courses

| | |
|---|---|
| 13-31 January: | 26 students, Technical University |
| 2 and 9 April: | 6 students, and |
| 15-16 April: | 8 students, Royal Veterinary and Agriculture University, |
| 21-22 April: | 5 students, and |
| 23-24 April: | 5 students from the Technical University in Lund Sweden. |

## 7.2 Neutron Radiography

The shielding facility was altered so that irradiated fuel rods of 4.5 m in length can now be radiographed. 14 uranium rods irradiated in the Halden reactor, and several minipins ramp tested in DR 3 were radiographed.

The pictures are made by means of X-ray films, but pictures were also made by means of plastic foils.

## 7.3 Pile Oscillator

The electronic part of the new oscillator is fully transistorized and easy to operate, but unfortunately the equipment is very difficult to calibrate and therefore this part is now being redesigned.

## 7.4 Germanium Detector

The contents of $^{137}Cs$ in 60 Dragon particles were measured to determine the burn-up. Iron samples from the corrosion test rigs were measured, and the thermal and fast neutron fluence determined. Further, several Co-foils were measured and the neutron flux calculated for the Isotope Laboratory.

## 7.5 Mössbauer Effect

The spectrometer was used to determine the heating up temperature of some soil samples. Temperatures were determined by a change of peak positions and line broadening.



# 8. Studies and Tasks

## 8.1 Economics of Nuclear Power

During 1974 an economic comparison was made between the three types of base load plants that can be expected to extend the present system of power plants during the mid-1980's; these are 1) a 900 MW(e) LWR, 2) a 600 MW(e) CANDU, 3) a 600 MW(e) coal/oilfired plant. The power plants were each treated individually, i.e. the existing system and the need for reserve capacity were not taken into account.

During 1975 we studied the significance of taking the complete system into account, including all power plants, the grid and the possibilities of supply from neighbouring systems. Some of the implications for the system of making a comparison between plant units of different sizes, and between units which might be of differing reliability, were investigated. Calculations were carried out for the Kraftimport area to obtain an estimate of the effects of introducing large units on the grid in the mid-1980's. As an integral part of this investigation, an evaluation was made of the significance of the strength of the electrical connections to neighbouring supply systems for the types of unit which, from the standpoint of economics and ensuring supplies, would be acceptable to the Kraftimport system. Furthermore, general calculations were carried out to investigate the relations between installed capacity, unit sizes, plant reliability factors, power demand conditions and security of supplies.

These calculations were carried out using two programs implemented for the purpose on the B-6700 computer: an original Belgian program, which can treat two collaborating systems, and a French program, which can treat an area comprising several collaborating systems.

An evaluation of published performance data for LWR power plants, as well as for coal/oil-fired plants, was carried out in order to obtain statistical data on plant availability and reliability for the two types of plants in various unit sizes. The data were slightly modified to give values relevant to plants envisaged started up in the mid-1980's.

Finally, a member of the group has participated in a working group on power plant economics set up by the Ministry of Commerce.

## 9. Publications

### 1. Risø Reports

Morten Lind
Elements of Automata Theory and the Theory of Markov Chains.
Risø Report No. 309

Erik Nonbøl
Development of a Dynamic Model of a BWR Nuclear Power Plant.
Part I: The Steady-State Model
(In print) Risø Report No. 335

Erik Nonbøl
Development of a Dynamic Model of a BWR Nuclear Power Plant
Part II: The Dynamic Model
(In print) Risø Report No. 336

### 2. Various works published

F. List
Grønlandsk malm kan gøre Danmark selvforsynende med energi. Berlingske Tidende, 15/10 1975

F. List
Udviklingen på det nukleare område i udlandet.
AEK's årsberetning 1974/75.

F. List
Atomenergien i 1974 "Ingeniør- og Bygningsvæsen"
4. april 1975.

F. List
Hurtige formeringsreaktorer. Bilag til "Kort Nyt"
18. december 1975.

F. List
Atomenergi og Kernekraftværker.
"Elektroteknikeren" 19. december 1975



P.S. Andersen
J. Würtz
A Film-flow Model for Annular Flow in an Annulus
May 1975.

J.-P- Bento et. al.
Comparison of the Draft of the U.S. Reactor Safety Study and the Swedish Urban Siting Study. S-483

H.E. Kongsø
REDIS, A Computer Program for System Reliability Analysis by Direct Simulation. IAEA-SM-195/17

D.J.Brown, A. Jensen, P.B.Whalley
Non-Equilibrium Effects in Heated and Unheated Annular Two-Phase Flow ASME-75-WA/HT-7

### 3. External Risø-M-Reports

P. la Cour Christensen
Users Mannual for the PWR-PLASIM model.
(1975) Risø-M-1757

The Department of Reactor Technology
Annual Progress Report
1 January - 31 December 1974
January 1975 Risø-M-1769

C.F. Højerup
Use of Thorium in a PWR
6 March 1975 Risø-M-1778

P.S. Andersen
Response to the SCNI ad hoc Group
Standard Problem Program
March 1975 (NORHAV-D-13). Risø-M-1785

### 4. Internal Risø-M-Reports

C.F. Højerup
Introduction of the Th-U233 cycle in the reactor codes at Risø, and some exercises performed at that occasion.
23 January 1975 Risø-M-1770

H.E. Kongsø and R. Korre Larsen
REDIS, A Computer Program for System Reliability Analysis by Direct Simulation. Program Description and Manual.
June 1975 — Risø-M-1781

Iqbal Ahmed
Determination of Heating Up Temperatures of Some Glozel (France) Soil Samples by the Use of Mössbauer Spectroscopy.
December 1975 — Risø-M-1832

## 5. Section of Reactor Engineering Reports (SRE)

Kurt Lauridsen
Notat vedr. strålingsniveauer på toppen af BWR med tryktank i forspændt beton
Januar 1975 — SRE-1-75

P.E. Becher
Statistik model for forspænding
18. februar 1975 — SRE-2-75

P.E. Becher
PFM-690
April 1975 — SRE-3-75

P.E. Becher
PEP 706
April 1975 — SRE-4-75

E. Bardram
Mødereferat Progressmøde no. 2527-1, Kredsløbsteknik
7. April 1975 — SRE-5-75

P.E. Becher
Idealiseret eksempel på probabilistisk analyse af forspændt beton-konstruktion
Maj 1975 — SRE-6-75

R. Korre Larsen, H.E. Kongsø
REDIS, A Computer Program for System Reliability Analysis by Direct Simulation
May 1975 — SRE-7-75



E. Bardram
Forskrifter og vejledning vedrørende licensering af kraftreaktorer i USA
Juni 1975 — SRE-8-75

F. List
NDT-PIE for HP 060 og 061 efter pre-cycle-bestråling.
Juni 1975 — SRE-9-75

E. Bardram
Kredsløbsteknik. Mødereferat Progressmøde no. 2527-3 (og 2)
3. september 1975 — SRE-10-75

E. Bardram
Kredsløbsteknik. Mødereferat Progressmøde no. 2527-4
September 1975 — SRE-11-75

E. Bardram
Kredsløbsteknik. Mødereferat Progressmøde no. 2527-5
1. oktober 1975 — SRE-12-75

H.E. Kongsø, Kurt Petersen
Abstract vedr. Abnormal Occurrencies 1972 og 1973
Oktober 1975 — SRE-13-75

E. Bardram
Kredsløbsteknik. Mødereferat Progressmøde no. 2527-6
24. oktober 1975 — SRE-14-75

P.E. Becher
Kredsløbsteknik. Mødereferat Progressmøde no. 2527-7
7. november 1975 — SRE-15-75

E. Bardram, K. Lauridsen, P.E. Becher
Kredsløbsbeskrivelse. Kølesystemet for en kogendevandsreaktor af typen General Electric BWR 6
November 1975 — SRE-16-75

H.E. Kongsø
Skematisk oversigt over BWR-transienter
13. november 1975 SRE-17-75

P.E. Becher
Referat af gruppemøde 13/11
17. november 1975 SRE-18-75

P.E. Becher
Ref. af kredsløbsgruppemøde no. 2527-8
19. november 1975 SRE-19-75

Kurt Hansen, H.E. Kongsø
Kredsløbsbeskrivelse, Nedluknings-køle-systemet for en kogendevandsreaktor af ty-pen General Electric BWR 6
26. november 1975 SRE-20-75

## 6. Reactor Physics Reports (RP)

Torben Petersen
Idrifttagning af Barsebäck 1 RP-1-75

G.K. Kristiansen
Calculation of fast flux in HP-rigs
April 1975. RP-2-75

Ib Misfeldt
2D IAEA Benchmark Problem
June 1975. RP-3-75

Ib Misfeldt
3D IAEA Benchmark Problem. 2nd order FEM calculation.
June 1975 RP-4-75

Ib Misfeldt
3D IAEA Benchmark Problem. 3rd order FEM calculation
June 1975 RP-5-75

G.K. Kristiansen
Statusrapport for Risø's reaktorfysiske arbejde til det 19. nordiske Reaktor-fysikmøde.
August 1975. RP-6-75



## 7. Section of Heat Transfer and Hydraulics Reports (SHH)

K. Ladekarl Thomsen
Description of the CONTAC Containment Code
May 1975 (MXA-2-105) (Commercial) SHH-1-75

K. Ladekarl Thomsen
Calculations carried out with the CONTAC Code compared with Marviken Test Results
May 1975 (MXA-2-205) (Commercial) SHH-2-75

F. Cortzen
Numerical Integration of First-order Differential Equations by One-step Methods
July 1975 SHH-3-75

Jens G. Munthe Andersen
H. Abel-Larsen
CORECOOL, Part I, Model Description
August 1975 (Commercial) SHH-4-75

CORECOOL, Part II, Program Description SHH-5-75
August 1975 (Commercial)

F. Cortzen
Comparison between the CONTACT-III and CONTEMPT-LT Containment Loss of Coolant Code
December 1975 SHH-7-75

## 8. NORHAV-reports

P.S. Andersen
Status of the Theoretical Foundations for Dynamic Subchannel Analysis including Results for One-dimensional Calculations
February 1975 — NORHAV-D-12

P.S. Andersen
Response to the CSNI ad hoc Group Standard Problem Program
March 1975 (Risø-M-1785) — NORHAV-D-13

H. Abel-Larsen
Small Breaks. A Review Based on Available Literature
May 1975 — NORHAV-D-14

P.S. Andersen
The Treatment of Slip in the Dynamic Subchannel Program Including Results for Critical Flow
October 1975 — NORHAV-D-15

## 9. Section of Dynamics Reports (DYN)

Erik Nonbøl
Users Manual to the Program "TURBPLANT"
(1975) — DYN-1-75

P. Skjerk Christensen
Udvælgelse af BWR-transienter egnede for analyse af arbejdsgangen ved behandling af en ABT.
(1975) — DYN-2-75

P. Skjerk Christensen
Xenontransienter i en een-dimensional Reaktor
(1975) — DYN-3-75



## 10. Staff of the Department of Reactor Technology

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